NON-PUBLIC?: N
ACCESSION #: 8710270236
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 of 5

DOCKET NUMBER: 05000410

TITLE: Reactor Scram on High Neutron Flux due to Personnel Error
EVENT DATE: 10/01/87 LER #: 87-058-00 REPORT DATE: 10/23/87

OPERATING MODE: 2 POWER LEVEL: 002

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Robert G. Randall, Supervisor Technical Support
TELEPHONE #: 315-349-2445

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SD COMPONENT: Z1 MANUFACTURER: 8040
REPORTABLE TO NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On October 1, 1987 at 1213 hours, Nine Mile Point Unit 2 experienced actuation of an Engineered Safety Feature (ESF), specifically, a high neutron flux scram on the Intermediate Range Monitors (IRM). At the time of the event, the plant was in the startup mode with reactor power at approximately 2.5% and the reactor mode switch in the "Startup/Hot Standby" position. Reactor pressure and temperature were approximately 505 psig and 477 degrees Fahrenheit respectively.

The root cause of this event was personnel error.

Corrective actions for this event are:

1. The individuals involved with this event were immediately instructed on the system characteristics and participated in the trip investigation and follow up report. The event was reviewed by the operators during shift turnovers.
2. A Training Modification Recommendation has been initiated to address this issue.
3. A work request (WR# 125604) has been written to investigate/repair the

condensate booster pump minimum flow valve and position transmitter.
4. A Temporary Change Notice has revised the Operating Procedure N2-OP-101A giving instructions to maintain steam loads within the capability of the condensate level control valve LV-137.
5. A Problem Report (PR# 7387) has been submitted to Engineering to evaluate level control valve LV-137. Engineering will make a determination as to whether the valve is undersized.
6. A Problem Report has been initiated to evaluate the vibration induced failure of the condensate minimum flow valve position transmitter.

(End of Abstract)

I. DESCRIPTION OF EVENT

On October 1, 1987 at 1213 hours, Nine Mile Point Unit 2 experienced actuation of an Engineered Safety Feature (ESF), specifically, a high neutron flux scram on the Intermediate Range Monitors (IRM). At the time of the event, the plant was in the startup mode with reactor power at approximately 2.5% and the reactor mode switch in the "Startup/Hot Standby" position. Reactor pressure and temperature were approximately 505 psig and 477 degrees Fahrenheit respectively.

Niagara Mohawk licensed operators were preparing to start feedwater pump 2FWS-P1A per Operating Procedure N2-OP-101A (Figure 1). A condensate booster pump was started (2CNM-P2B) and flow to the reactor vessel was being regulated through the condensate level control valve LV-137. The Reactor Water Cleanup System (RWCU) was in the full reject mode at approximately 175 gpm. Due to a feed flow/steam load mismatch, reactor water level slowly started to decrease. The operators immediately noticed that the level control valve was full open, and that additional inventory needed to be added to the vessel before reactor water level decreased to the low level alarm setpoint. The operators, believing they could increase feed flow to the vessel by throttling flow around the feedwater pump, opened MOV122 (feedwater pump bypass valve). This opening of the valve (MOV122) caused a reactor overfeed situation which produced a cold water power excursion and subsequent IRM high level scram. A high level turbine trip also occurred at approximately the same time.

II. CAUSE OF EVENT

A root cause analysis for this event has been completed per Site Supervisory Procedure S-SUP-1, "Root Cause Analysis Program". The root cause has been determined to be personnel error. The operator assumed

he could throttle flow around the feedwater pump, when he opened MOV122, to slowly increase reactor water level. What he failed to realize is that the logic for this valve is such that if an open signal is received, the valve's position goes full open and does not allow the valve to be throttled. This opening of the valve (MOV122) caused a reactor overfeed situation which produced a cold water excursion and IRM high level scram.

Contributing to the root cause of this event is procedural deficiency and instrument failure.

Procedural Deficiency - The procedure the operators were adhering to (N2-OP-101A) did not provide direction for controlling steam loads within the capability of the level control valve LV-137 with cleanup reject at 175 gpm. Therefore, when the operator experienced a condition outside the operating procedure, (feed flow/steam load mismatch) he assessed the condition and made a determination to throttle open MOV122. The basis for this determination was that reactor water level was decreasing and level control valve LV-137 was full open.

Instrument Failure - When the operator started condensate booster pump 2CNM-P2B, he observed a false annunciation (minimum flow valve position indicator read closed when in fact the valve was open) due to a failed transmitter. This failure was not the cause of the scram, but is believed to have distracted the operator during the course of the event prior to the scram. The failed transmitter has been attributed to vibration on the minimum flow valve line.

III. ANALYSIS OF EVENT

The reactor scram which occurred as a result of the Intermediate Range Monitor (IRM) trip is a conservative action and poses no adverse safety consequences at any reactor power level. The event did not in any way adversely affect any other safety systems or the operators ability to achieve safe shutdown. All systems operated as designed and no other engineered safety features were actuated.

IV. CORRECTIVE ACTIONS

1. The individuals involved with this event were immediately instructed on the system characteristics and participated in the trip investigation and follow-up report. The event was reviewed by the operators during shift turnovers.

2. A Training Modification Recommendation (TMR# 02-87.238) has been initiated to address this issue. This event will be reviewed by all licensed operators during ongoing operator training.

3. A work request (WR# 125604) has been written to investigate/repair the condensate booster pump minimum flow valve and position transmitter. It has been determined that the position transmitter needs to be replaced. Engineering has been in contact with the vendor to supply transmitter replacements.

4. A Temporary Change Notice has been written which revised the Operating Procedure N2-OP-101A and added specific instructions to maintain steam loads within the capability of the level control valve LV-137.

5. A Problem Report (PR# 7387) has been submitted to Engineering to evaluate level control valve LV-137 and to make a determination as to whether there is a possibility the valve is undersized.

6. A Problem Report has been initiated to evaluate the vibration induced failure of the condensate minimum flow valve position transmitter.

V. ADDITIONAL INFORMATION

Identification of Components Referred to in this LER

IEEE 803 IEEE 805
Component EIIS Funct System ID

MSIV N/A JC
Feedwater Pump P SJ
Bypass Valve V SD
Condensate Booster Pump P SD
Level Control Valve LCV SD
Reactor Water Cleanup System N/A CE
Intermediate Range Monitor N/A IG
Position Indicator (Transmitter) ZT SD

There have been no previous similar events.

FIGURE 1

FIGURE OMITTED - NOT KEYABLE (DIAGRAM)

NMP26479

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

301 PLAINFIELD ROAD
SYRACUSE, NY 13212

THOMAS E. LEMPGES
VICE PRESIDENT -- NUCLEAR GENERATION

October 23, 1987

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 87-58

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 87-58 Is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported."

A 10CFR50.72 report for this event was made at 1340 hours on October 1, 1987.

This report was completed in the format designated in NUREG-1022, Supplement No. 2, dated September 1985.

Very truly yours,

/s/ Thomas E. Lempges

Thomas E. Lempges
Vice President
Nuclear Generation

TEL/SCN/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

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